

NON-PUBLIC?: N
ACCESSION #: 9002120339
LICENSEE EVENT REPORT (LER)

FACILITY NAME: RIVER BEND STATION PAGE: 1 OF 11

DOCKET NUMBER: 05000458

TITLE: Reactor Scram Due to Main Generator Exciter Brush Failure
EVENT DATE: 08/25/88 LER #: 88-018-04 REPORT DATE: 01/31/90

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: D SYSTEM: EA COMPONENT: 27 MANUFACTURER: W120
D TL EXC GO84
B LD CMP 1076
REPORTABLE NPRDS: Y
N
N
SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At 1232 on 8/25/88 with the unit at 100 percent power (Operational Condition 1), the reactor automatically scrammed due to a turbine control valve fast closure caused by a loss of main generator field excitation resulting in automatic main generator and turbine trips. Immediately following the scram, reactor pressure spiked to a peak between 1100 and 1117 psig causing the five low-low set safety relief valves to cycle per design. The turbine bypass valves opened as required and the reactor recirculation pumps transferred to slow speed per design. Reactor water level initially decreased to +4 inches as indicated by the wide range instruments due to the reactor pressure spike. The high pressure core spray (HPCS) and reactor core isolation cooling (RCIC) systems injected

as a result of a spurious low reactor water level 2 signal caused by a hydraulic perturbation in the reactor water level instrument reference lines. As a result of the feedwater flow continuing (due to the "A" feedwater control valve being in the manual mode at 50 percent open) in conjunction with the HPCS and RCIC injections, reactor water level rapidly increased to level 8 causing the HPCS injection valve and the RCIC steam supply valve to close and the reactor feedwater pumps to trip per design.

There was no significant adverse impact on the safe operation of the plant or to the health and safety of the public as a result of this event since the reactor scram placed the unit in the safe shutdown condition.
END OF ABSTRACT

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REPORTED CONDITION

At 1232 on 8/25/88 with the reactor at 100 percent power (Operational Condition 1), the main turbine (*TRB*) automatically tripped due to a generator (TG) trip on loss of field excitation, resulting in an automatic reactor scram due to the turbine control valve (*FCV*) fast closure.

Prior to the reactor scram, one of the main generator exciter brushes was identified as sparking. The control room exciter field volt meter (*EI*) was showing erratic readings between 33 to 70 DC volts while the voltage regulator (*RG*) was in the manual mode. Maintenance personnel were finalizing work details in preparation for replacing the worn exciter brushes when the main generator tripped on loss of field excitation.

Concurrent with the generator/turbine trip, the reactor water recirculation (*AD*) pumps (P) automatically transferred to the low frequency motor generator (LFMG) sets (*MG*) (slow speed) per design upon receiving an end-of-cycle recirculation pump trip (EOC RPT) signal.

Immediately following the scram reactor pressure spiked to a peak between 1100 and 1117 psig causing the five low-low set safety relief valves (SRVs) (RV) to cycle per design. The turbine bypass valves (PCV) also opened as required. The at-the-controls (ATC) operator maintained control of reactor pressure via use of the turbine bypass valves.

Reactor water level initially decreased due to the collapse of steam voids as a result of the reactor pressure spike. The lowest actual water level reached was +11 inches, +10 inches and +6 inches as indicated by the "A", "B", and "C" channel narrow range instruments (*LT*),

respectively. The lowest wide range water level indication was +4 inches. However, the plant computer (*CPU*) showed evidence of a hydraulic perturbation on the wide range level instrumentation (*LT*) resulting in a low level spike in excess of -29 inches.

During the voltage transient caused by the generator trip, non-safety related 4.16 KV switchgear (*SWGR*) 1NNS-SWG1A failed to transfer (fast and slow) from normal station service transformer (*XPT*) 1STX-XNS1C to preferred station service transformer (*XPT*) 1RTX-XSR1C as a result of circuit breaker (*52*) 1NNS-ACB007 failing to close. This resulted in a loss of power to the high pressure core spray (HPCS) (*BG*) safety-related 4.16 KV bus (*EB*) 1E22*S004 and the non-safety related 4.16 KV bus (*SWGR*) 1NNS-SWG1C. Non-safety related 4.16 KV bus (*SWGR*) 1NNS-SWG1B also failed to fast transfer but successfully completed a slow transfer upon automatic closure of circuit breaker (*52*) 1NNS-ACB015.

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As a result of the loss of power to the 1E22*S004 bus, the division III standby service water (SSW) (*BI*) pump (*P*) (1SWP*P2C) automatically started from a low normal service water (NSW) system (KG) signal caused by the loss of power to the initiating instrumentation.

Additionally, the HPCS diesel generator (EK) received an automatic initiation signal due to the undervoltage condition on the 1E22*S004 bus. The HPCS diesel generator successfully started and its output breaker (52) automatically closed, restoring voltage to the bus per design.

Due to the loss of power to 1NNS-SWG1A as described above, the turbine plant component cooling water (TPCCW) system (*KB*) pumps (*P*) 1CCS-P1A and 1CCS-P1C tripped. Operations personnel verified that 1CCS-P1B automatically started. However, this pump alone was insufficient to maintain proper cooling to the plant instrument air system (*LD*) compressors (*CMP*). As a result, the instrument air compressors tripped on high temperature. Operations personnel restarted a second TPCCW pump after restoring power to 1NNS-SWG1A. The instrument air compressors were then restarted and continued to operate properly. The lowest instrument air header pressure indicated was 80 psig. Operations personnel reported no unanticipated cycling of any air operated valves (*V*) or dampers (*DMP*).

Additionally, power to reactor protection system (RPS) (*JC*) bus "A" (*EC*) was lost. RPS bus "B" continued to operate with power supplied from the "B" RPS M-G (*MG*) set. RPS bus "A" was manually transferred to alternate supply, restoring power.

The loss of power to RPS bus "A" also resulted in an automatic.

initiation of the standby gas treatment (*SGTS*) (*BH*) and annulus mixing (AM) (*VC*) systems and an automatic trip of the annulus pressure control (APC) system (*VC*). The three systems responded as designed upon the loss of RPS power.

A spurious high drywell pressure alarm (PA) also actuated as a result of the loss of power to RPS bus "A". The emergency response and information system (ERIS) computer (CPU) recordings verified that actual drywell pressure did not exceed 0.5 psid. The high drywell pressure trip setpoint is 1.68 psid.

The HPCS and reactor core isolation cooling (RCIC) (BN) systems received an automatic initiation signal and injected. These initiation signals resulted from a spurious reactor water level 2 signal caused by the hydraulic perturbation on the wide range reactor water level instrumentation previously described. The controller (PMC) for the "A" feedwater (LC) control valve (FCV) was in the manual mode at 50 percent position. As a result of feedwater flow continuing and the HPCS and RCIC injections, reactor water level

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rapidly increased to level 8 causing the HPCS injection valve (*INV*) and the RCIC steam supply valve (ISV) to automatically close and the three reactor feedwater pumps (P) to trip per design. The HPCS and RCIC systems responded as designed and injected to the reactor vessel for approximately 31 and 30 seconds, respectively. A notification of unusual event (NOUE) was entered at 1253 hours based on the emergency core cooling system (ECCS) injection into the reactor vessel. The NOUE was subsequently terminated at 1320 on 8/25/88.

At approximately one hour after the reactor scram, plant personnel reported that they observed heat radiating from the HPCS injection line upstream of the HPCS injection valve. The control room was notified and further investigation and evaluation began.

INVESTIGATION

An investigation of the main generator trip determined that deterioration of the exciter brushes led to the loss of exciter field voltage resulting in the subsequent generator trip. The root cause of the exciter brush failure was determined to be a deficiency in the preventive maintenance procedure. The investigation revealed that no specific requirement had been established within the procedure as to when to replace the exciter brushes. The exciter brushes were replaced prior to initiating subsequent startup procedures. A thorough investigation of the generator

and turbine revealed no other damage. An investigation into the cause of the HPCS and RCIC injections revealed the source of the initiation signals to be a spurious hydraulic perturbation in the reactor water level instrumentation reference lines and not an actual low reactor water level 2 signal. The hydraulic perturbation was caused by the 100 percent turbine trip induced reactor steam dome pressure spike. The pressure spike was immediately transmitted to the four reactor level reference lines located near the top of the reactor vessel but was not immediately sensed by the narrow and wide range variable line taps located lower in the reactor vessel. Using the Sequence of Events Report from the plant process computer, it was determined that a HPCS low water level signal was received on channels "G", "L", and "R" for 20, 38 and 42 milliseconds, respectively. In addition, the RCIC initiation signals occurred at the same time. The instruments used to measure reactor water level for these initiation logics are Rosemount type 1154 transmitters which are fast acting instruments with no electronic dampening. As a result, the instruments caused the automatic initiation based upon sensing the pressure spike of very short duration.

Using ERIS data, HPCS and RCIC instrumentation showed significant spikes 300 milliseconds after the scram. ERIS monitors these instrument signals in 100 millisecond intervals and showed reactor

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water level spike magnitudes of -19, -29, -16, and -4 inches. The magnitude of the signals recorded by ERIS cannot be considered absolute due to the ERIS sample rate time intervals.

Additional evidence which shows that an actual low reactor water level 2 was not reached includes the fact that signals were not received on the following reactor water level instruments: 1) RPS level 3 scram, 2) nuclear steam supply shutoff system (JM) (NSSSS) level 2 balance of plant (BOP) and reactor water cleanup system (RWCU) (CE) equipment isolation, 3) anticipated transient without scram (ATWS) level 2 trip and 4) feedwater low level annunciator/computer points. The above signals are generated by Rosemount 1152 transmitters which have built in electronic dampening to slow the response of the instruments. These instruments also share common reference lines with the Rosemount type 1154 transmitters. Therefore, it is concluded that the initiation signals for HPCS and RCIC were generated as a result of the lack of electronic dampening in the Rosemount 1154 transmitters.

To support restart, a preliminary evaluation was conducted to determine the impact of the elevated temperatures in the HPCS injection line. Temperatures were taken to determine the profile and maximum temperatures

that the system had seen. This conservative evaluation concluded that there was no impact on the integrity of the piping system and in fact, the HPCS system could sustain at least one additional thermal transient of this type without affecting the designed life of the piping. A best-estimate maximum temperature profile for the HPCS injection pipe was calculated using RELAP5. This best-estimate temperature profile shows that the initial temperature profile used to assess piping integrity was conservative.

Additionally, samples of the water in the injection line confirmed that the water did come from the reactor vessel.

It is believed that reverse flow through the HPCS injection check valve 1E22*A0VF005 was caused by the following events:

1. Check valve 1E22*A0VF005 did not seat completely due to throttling of the flow upon termination of HPCS injection when high reactor water level was reached. As 1E22*M0VF004 was throttled closed, the pressure between the MOV and 1E22*A0VF005 would be expected to slowly decrease to reactor pressure and then remain constant. Since the pressure was the same both upstream and downstream of the check valve and no reverse flow occurred (i.e., pipe is full of water and no flow path available), the only closing force on the valve was gravity. It is believed that the slow flow reduction and the lack of a large closing force resulted in the valve not seating properly.

During RF-2, 1E22*A0VF005 passed its initial LLRT. The valve was then completely disassembled per MWO 126282 and inspected for

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damage. Dimensions of selected internal components were recorded. No damage or unusual conditions were noted; however, all seating surfaces were reworked prior to reassembly. The valve then passed a second LLRT. The as found condition of the valve supports the theory that the valve did not seat completely due to throttling of HPCS flow.

2. The HPCS injection valve 1E22MOVFOO4 cycled as shown by ERIS computer data.

3. Minor leakage occurred through one or more of the following valves while 1E22*M0VF004 was open:

- a. 1E22*VF024 HPCS pump discharge check valve and 1E22*M0VF012 HPCS pump minimum flow valve.

Any leakage through 1E22*VF024 is assumed to have been minor. This assumption is supported by the closing of 1E22*M0VF012. The closing of the minimum flow valve indicates that pressure upstream of the check valve decreased below the closing setpoint for 1E22*M0VF012. Major leakage or failure of the check valve would have resulted in the minimum flow valve remaining open (i.e., high discharge pressure and low flow). ERIS data for HPCS system pressure (1E22*PTN051, ERIS point 1E22-N0002) supports this assumption.

b. 1E22*M0VF023 HPCS test return to suppression pool.

1E22M0VF023 received an LLRT during RF-2. The valve passed the initial LLRT and no rework was required. The results of the LLRT during RF-2 supports the conclusion that any leakage which may have occurred was minor.

Other potential leakage paths are through 1E22*M0VF010 and F011 HPCS test return to the condensate storage tank or 1E22*RVF035 HPCS discharge relief valve.

It does not appear that the actual leakage path can be determined with certainty. The most probable leakage path is a combination of minor leaks through several valves.

Evaluation of the effects of a possible water hammer on the HPCS piping as it relates to this event has been performed. GSU has evaluated two conditions where a water hammer could be postulated to have occurred in the HPCS piping following the reactor scram. The first is during automatic start up of the HPCS system after the scram. If a void was present during system start up, a water hammer could have occurred. A water hammer under these conditions would have been characterized by large fluctuations in system flow and pressure. A review of the ERIS traces of HPCS flow and pressure indicate that no water hammer occurred during the system initiation.

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The second time that a water hammer can be postulated to have occurred is when the injection valve (1E22*M0VF004)(*INV*) stroked after the HPCS pump was shutdown. In this case, the ERIS traces do not provide any useful data that can be used to determine whether or not a water hammer actually occurred. The HPCS flow indicator (1E22*FIR603) (*FI*) was indicating zero flow and did not change. This instrument, however, is a differential pressure type. Under reverse flow conditions, it is expected that this instrument would "peg low" and thus the ERIS trace

would not be expected to change. The HPCS pump discharge pressure indicator (1E22*PIR601) (*PI*) located upstream of the pump discharge check valve (1E22*VF024) (*V*), did not show any pressure fluctuations. Due to its location, this pressure indicator would see pressure only if the pump discharge check valve was stuck open. Although the lack of pressure fluctuation indications does not provide any information as to whether or not a water hammer actually occurred, it does provide evidence that the pump discharge check valve operated properly.

Following the HPCS pipe thermal transient, a walkdown of the HPCS system was conducted. The HPCS piping and supports were visually inspected for damage and no damage was noted. Surface and volumetric examinations were also performed on those welds identified as having experienced the highest thermal stresses. These examinations revealed no unacceptable indications and, thus, verified the structural integrity of the system.

Based upon the above, no degradation to the system occurred. Additionally, a turbine trip from full power reactor scram occurred on 9/6/88 with a HPCS system injection as reported in LER 88-021. Following this scram, the HPCS pipe thermal transient experienced during the first reactor scram did not recur.

An evaluation was performed using RELAP5/BLAZER to determine the plausibility of the two possible water hammer events. Results of the evaluation show that both events were possible. The impact of these events was determined by comparing the calculated loads from the new postulated events with the design basis loads. The original design water hammer loadings are as follows:

1. Pump start with air void in discharge line,
2. Check valve slam from pump stop, and
3. HPCS system in test mode and then transfer to injection mode.

Water hammer loads for the events, both actual and postulated, associated with reactor scram 88-04 have been analyzed using the RELAP5 and BLAZER programs. The RELAP5/BLAZER programs have been qualified for use in water hammer evaluation using approved procedures. The results of the analysis of the actual loadings that

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occurred were compared with the original design basis water hammer loadings. This comparison showed that the original design basis water

hammer loadings were greater than any water hammer loadings that actually occurred after this scram. Therefore, the conclusion that there was no damage to the piping system as stated above is confirmed.

The analysis for the evaluation of the postulated event of a HPCS pump start with a void created by reactor water back flow yielded higher transient loads than the loadings used in the original pipe stress analysis. A pipe stress analysis was subsequently performed and determined that the stress levels in the pipe would be below allowable stress limits. Eight of the supports would exceed emergency condition allowable loads and stresses. All of these supports meet the requirements for faulted condition allowable loads and stresses for all components. The loading on the discharge nozzle of the HPCS pump would increase beyond the qualified allowable loads. Review of the qualification report and a conservative comparison of postulated loads to current allowables shows that the stresses in the highest stressed points in the pump would be increased to approximately the tensile strength of the material. The conservative nature of this comparison leads to the conclusion that the pump materials would not have failed. It is concluded that the piping, supports and system would have been able to fulfill their required safety functions if this event had occurred. Note that the event being postulated in this analysis did not occur after scram 88-04, as stated before, the thermal transient event itself did not occur after the scram on 9/6/88.

Additionally, an evaluation has been performed for the V event (see WASH 1400) for high/low pressure interface and the impact on the system piping as it relates to this event. The WASH 1400 V event, Interfacing System LOCA, is the rupture of low pressure piping initiated by the failure of components isolating the high pressure reactor coolant system from the low pressure piping. The high/low pressure boundary of the HPCS system is the pump suction connection. The pump discharge check valve (1E22*VFO24) is on the high pressure side of this interface and acts as one of the barriers between the low pressure and high pressure piping.

HPCS pump discharge pressure is provided to ERIS by a transmitter between the pump discharge nozzle and the discharge check valve. ERIS traces are available which provide conclusive evidence that the suction piping was not overpressurized during the transient.

The final evaluation of the stresses that would have been experienced by the low pressure piping if it had been pressurized to reactor pressure indicates that the piping would not have ruptured.

Based upon the above discussion, the transient experienced by the HPCS piping following the reactor scram does not impact the V event

probabilities at River Bend Station.

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During the transient which occurred concurrent with the reactor scram, 1NNS-SWG1A failed to transfer from normal station service transformer 1STX-XNS1C to preferred station service transformer 1RTX-XSR1C. An investigation has revealed the following events led to the failure to transfer. As a result of the flashover of the main generator exciter brushes, the main generator output voltage decreased. This caused the voltage at the Fancy Point Substation to decrease and undervoltage relay (*59*) 59R-1NNSA08 to de-energize. This undervoltage relay must be energized in order for circuit breaker 1NNS-ACB07 to close to complete either a fast or slow bus transfer. Undervoltage relay 59R-1NNSA08 monitors voltage on the off-site power line (EA) from the Fancy Point Substation which supplies preferred station service transformer 1RTX-XSR1C and ensures that sufficient voltage is available from the off-site power source prior to allowing the automatic transfer to take place.

When the generator output circuit breaker opened, the voltage at Fancy Point Substation began to stabilize. However, 10 cycles after the generator output breakers opened, relay (*62*) 62XG-1SPGN07 energized by design and disabled the fast closure portion of the 1NNS-ACB07 closing circuit. By the time the voltage at Fancy Point stabilized and undervoltage relay 59R-1NNSA08 re-energized, relay 62XG-1SPGN07 had disabled the fast closure portion of 1NNS-ACB07.

In order for 1NNS-ACB07 to close on a slow transfer, relay (94) 94B-1NNSA08 must be energized. Relay 94B-1NNSA08 is energized (after a short time delay) when both undervoltage relays (*27*) 27-1-1NNSA17 and 27-2-1NNSA17 sense a low voltage on the 1NNS-SWG1A bus. According to Operations personnel on duty, annunciator 0067 on panel 1H13-P808 had alarmed. This indicates that only one of the two 27 relays operated properly. Hence, 1NNS-ACB07 was also prevented from closing on a slow transfer. Additional evidence that relay 94B-1NNSA08 was not energized is that time delay relay (*62*) 62-1NNSA08 which activates 94B-1NNSA08 did not indicate a target, and the process computer point which is also activated by the 62 relay did not operate. The 27-1 and 27-2 relays addressed in this evaluation are not used in safety related applications at River Bend Station (RBS).

Inspections of the 27-1 and 27-2 relays revealed no apparent problems although the contacts were slightly dirty. Problems have occurred in the past from dirty or oxidized contacts on these undervoltage relays. The current preventive maintenance (PM) frequency is 24 months per the

vendor's recommendations. The undervoltage relays which were suspected of not properly operating were replaced. The remaining relays and timers in the transfer circuit were successfully tested.

An evaluation of the loss of power to RPS bus "A" revealed that the motor generator set output breaker tripped as a result of the voltage and frequency transients that were generated concurrent with the exciter brush failure. In addition to providing overcurrent

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protection, the M-G set output breaker (*52*) will trip on overvoltage or underfrequency. The RPS bus is further protected by two EPA breakers (*52*) in series with the M-G set breaker. The EPA breakers will also trip on overvoltage, undervoltage, or underfrequency.

An analysis of the output breaker trip circuit revealed that actuation of the overvoltage relay (*59*) causes the circuit breaker undervoltage trip device (*27*) to trip the breaker and energize relay 3K, which energizes relay 2K, which is then sealed in through the reset pushbutton. Neither the underfrequency relay nor the breaker overcurrent function will energize the 2K relay requiring the reset button to be depressed prior to resetting the output breaker. Therefore, it was determined that the M-G set output breaker had tripped due to an overvoltage condition.

It cannot be determined conclusively from the evidence available if the M-G set breaker tripped prior to the EPA breaker trip or as a result of it. Since all three breakers are in series, the resulting effect on the RPS bus would have ultimately been the same.

A review of previous LERs submitted by River Bend Station revealed no previous reactor scrams as a result of a failure of the main generator exciter or brushes.

CORRECTIVE ACTION

To assure that the main generator exciter brushes are replaced before deteriorating to a point which arcing occurs, the preventive maintenance procedure is being revised to establish specific wear criteria at which the brushes are to be replaced. The required procedure changes have been completed.

During the forced outage resulting from the reactor scram, maintenance personnel were trained by the vendor representative as to when and how to change the main generator exciter brushes. Following replacement of the brushes, an inspection was performed at 1800 rpm before synchronizing the

generator to offsite power (*FK*) to ensure proper operation of the exciter brushes, holders, and collector rings. No abnormal conditions were observed.

An engineering evaluation of the HPCS and RCIC initiation logic has been conducted to determine appropriate corrective actions for preventing unnecessary level 2 initiations caused by 100 percent turbine trip induced reactor water level instrument spikes. All eight Rosemount 1154 transmitters installed at River Bend Station which provide a trip function (including the transmitters which initiated the HPCS and RCIC systems reported in this LER) were modified during the second refueling outage. This modification included replacement of the amplifier circuit board and the calibration circuit board. This modification provided similar electronic dampening features as those contained in the Rosemount 1152 transmitters.

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The undervoltage relays which led to the failure to transfer of switchgear INNS-SWG1A have been replaced. The current preventive maintenance tasks for cleaning the undervoltage relay contacts are scheduled on a 24 month frequency per the vendor's recommendations. Maintenance has revised these tasks
o require a 6 month frequency.

As noted in the "Investigation" section, all analysis has been completed for the effects of reactor water entering the HPCS injection line. The analysis shows that all events that actually happened had no adverse affect on the piping system. Even with the postulated event of a pump start with a void created by reactor water back flow, the system would have been capable of performing its safety function. Based on this information and the implementation of the instrument modifications to prevent spurious actuations of HPCS, no additional corrective action is required.

SAFETY ASSESSMENT

There was no significant adverse impact on the safe operation of the plant or the health and safety of the public as a result of this event since the reactor scram placed the unit in the safe shutdown condition. The HPCS and RCIC systems, while actuation was unnecessary, did function properly to provide reactor water makeup and all other automatic safety system actions performed as designed.

NOTE: Energy Industry Identification System Codes are identified in the text as (XX).

ATTACHMENT 1 TO 9002120339 PAGE 1 OF 1

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U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Gentlemen:

River Bend Station - Unit 1
Docket No. 50-458

In accordance with 10CFR50.73, please find enclosed Licensee Event Report No. 88-018, Revision 4, "Reactor Scram Due to Main Generator Exciter Brush Failure". This final report is being submitted to inform you of the results of GSU's evaluation of the effects of water hammer on the HPCS piping.

Sincerely,

J.E. Booker
Manager-River Bend Oversight
River Bend Nuclear Group

JEB/TFP/RGW/DCH/MAS/pg

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